

ATTACHMENT 7

Westinghouse Report WCAP-15203,
Catawba Unit 1 Heatup and Cooldown Limit Curves for Normal
Operation Using Code Case N-640, Revision 1

Westinghouse Non-Proprietary Class 3



WCAP-15203

Revision 1

**Catawba Unit 1
Heatup and Cooldown
Curves for Normal
Operation Using Code Case
N-640**



Westinghouse Electric Company LLC

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-15203, Revision 1

**Catawba Unit 1
Heatup and Cooldown Limit Curves
for Normal Operation Using Code Case N-640**

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PREFACE

This report has been technically reviewed and verified by:

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Revision 1:

An error was detected in the "OPERLIM" Computer Program that Westinghouse uses to generate pressure-temperature (PT) limit curves. This error potentially effects the heatup curves when the 1996 Appendix G Methodology is used in generating the PT curves. It has been determined that WCAP-15203 Rev. 0 was impacted by this error. Thus, this revision provides corrected curves from WCAP-15203 Rev. 0.

Note that only the heatup curves and associated data point tables have changed. The cooldown curves and data points remain valid and were not changed.

TABLE OF CONTENTS

LIST OF TABLES.....	iii
LIST OF FIGURES.....	iv
EXECUTIVE SUMMARY.....	v
1 INTRODUCTION.....	1
2 PURPOSE.....	2
3 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS.....	3
4 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE.....	11
5 HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES.....	20
6 REFERENCES.....	30
APPENDIX A: TECHNICAL BASIS FOR REDUCED REACTOR VESSEL FLANGE REQUIREMENT.....	A-0

LIST OF TABLES

Table 1	Summary of the Vessel Surface, 1/4T and 3/4T Fluence Values used for the Generation of the 34 EFPY Heatup/Cooldown Curves	12
Table 2	Integrated Neutron Exposure of the Catawba Unit 1, McGuire Unit 2, and Watts Bar Unit 1 Surveillance Capsules Tested To Date	12
Table 3	Measured 30 ft-lb Transition Temperature Shifts of all Available Surveillance Data	14
Table 4	Reactor Vessel Beltline Material Unirradiated Toughness Properties	15
Table 5	Calculation of Chemistry Factors using Catawba Unit 1 Surveillance Capsule Data	16
Table 6	Summary of the Catawba Unit 1 Reactor Vessel Beltline Material Chemistry Factors ...	17
Table 7	Summary of the Calculated Fluence Factors used for the Generation of the 34 EFPY Heatup and Cooldown Curves	17
Table 8	Calculation of the ART Values for the 1/4T Location @ 34 EFPY	18
Table 9	Calculation of the ART Values for the 3/4T Location @ 34 EFPY	18
Table 10	Summary of the Limiting ART Values Used in the Generation of the Catawba Unit 1 Heatup/Cooldown Curves	19
Table 11	34 EFPY 60°F/hr. Heatup Curve Data Points Using 1996 App. G (without Uncertainties for Instrumentation Errors)	26
Table 12	34 EFPY 80°F/hr. Heatup Curve Data Points Using 1996 App. G (without Uncertainties for Instrumentation Errors)	27
Table 13	34 EFPY 100°F/hr. Heatup Curve Data Points Using 1996 App. G (without Uncertainties for Instrumentation Errors)	28
Table 14	34 EFPY Cooldown Curve Data Points Using 1996 App. G (without Uncertainties for Instrumentation Errors)	29

LIST OF FIGURES

Figure 1	Geometry of the Upper Head/Flange Region of a Typical Westinghouse Four Loop Plant Reactor Vessel	8
Figure 2	Crack Driving Force as a Function of Flaw Size: Outside Surface Flaw in the Closure Head to Flange Region Weld	9
Figure 3	Determination of Boltup Requirement, Using K_{Ic}	10
Figure 4	Catawba Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 60°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors).....	22
Figure 5	Catawba Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 80°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors).....	23
Figure 6	Catawba Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors).....	24
Figure 7	Catawba Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors).....	25

EXECUTIVE SUMMARY

This report provides the methodology and results of the generation of heatup and cooldown pressure temperature limit curves for normal operation of the Catawba Unit 1 reactor vessel. These curves were generated based on the latest available reactor vessel information (Capsule V analysis, WCAP-15117^[1] and the latest Pressure-Temperature (P-T) Limit Curves from WCAP-15118^[2]).

The Catawba Unit 1 heatup and cooldown pressure-temperature limit curves have been updated based on the use of the ASME Code Case N-640^[3], which allows the use of the K_{Ic} methodology, and a justification to lower the reactor vessel flange temperature requirement.

1 INTRODUCTION

Heatup and cooldown limit curves are calculated using the adjusted RT_{NDT} (reference nil-ductility temperature) corresponding to the limiting beltline region material of the reactor vessel. The adjusted RT_{NDT} of the limiting material in the core region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties, estimating the radiation-induced ΔRT_{NDT} , and adding a margin. The unirradiated RT_{NDT} is designated as the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (normal to the major working direction) minus 60°F.

RT_{NDT} increases as the material is exposed to fast-neutron radiation. Therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the unirradiated RT_{NDT} (IRT_{NDT}). The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steels. The Nuclear Regulatory Commission (NRC) has published a method for predicting radiation embrittlement in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials."^[4] Regulatory Guide 1.99, Revision 2, is used for the calculation of Adjusted Reference Temperature (ART) values ($IRT_{NDT} + \Delta RT_{NDT} + \text{margins for uncertainties}$) at the 1/4T and 3/4T locations, where T is the thickness of the vessel at the beltline region measured from the clad/base metal interface. The most limiting ART values are used in the generation of heatup and cooldown pressure-temperature limit curves.

2 PURPOSE

The Duke Power Company contracted Westinghouse to regenerate the 34 EFPY heatup and cooldown curves documented in WCAP-15118^[2] using K_{Ic} in place of K_{IR} for the calculation of the stress intensity factors. The heatup and cooldown curves from WCAP-15118 were generated without margins for instrumentation errors and included a hydrostatic leak test limit curve from 2485 to 2000 psig and pressure-temperature limits for the reactor vessel flange regions per the requirements of 10 CFR Part 50, Appendix G^[5] (as modified in Section 3.3 of this report).

The purpose of this report is to document the generation of new 34 EFPY P-T limit curves utilizing the K_{Ic} methodology^[3]. The P-T curves are developed with the identical adjust reference temperature (ART) values used in WCAP-15118. In addition, this report provides justification for relaxing the reactor vessel flange temperature requirement of Appendix G to 10CFR Part 50 based on the use of K_{Ic} methodology rather than the K_{IR} methodology. The use of K_{Ic} and relaxation of the reactor vessel flange temperature requirement will add substantial pressure margin to the heatup and cooldown curves documented in WCAP-15118. This increase in allowable pressure is presented in Section 5 of this report.

3 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

3.1 Overall Approach

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I , for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{Ic} , for the metal temperature at that time. K_{Ic} is obtained from the reference fracture toughness curve, defined in Code Case N-640, "Alternative Reference Fracture Toughness for Development of PT Limit Curves for Section XI"^[3 & 6] of the ASME Appendix G to Section XI. The K_{Ic} curve is given by the following equation:

$$K_{Ic} = 33.2 + 20.734 * e^{[0.02(T - RT_{NDT})]} \quad (1)$$

where,

K_{Ic} = reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}

This K_{Ic} curve is based on the lower bound of static critical K_I values measured as a function of temperature on specimens of SA-533 Grade B Class1, SA-508-1, SA-508-2, SA-508-3 steel.

3.2 Methodology for Pressure-Temperature Limit Curve Development

The governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C * K_{Im} + K_{It} < K_{Ic} \quad (2)$$

where,

K_{Im} = stress intensity factor caused by membrane (pressure) stress
 K_{It} = stress intensity factor caused by the thermal gradients
 K_{Ic} = function of temperature relative to the RT_{NDT} of the material
 C = 2.0 for Level A and Level B service limits
 C = 1.5 for hydrostatic and leak test conditions during which the reactor core is not critical

For membrane tension, the corresponding K_I for the postulated defect is:

$$K_{Im} = M_m \times (pR_i / t) \quad (3)$$

where, M_m for an inside surface flaw is given by:

$$\begin{aligned} M_m &= 1.85 \text{ for } \sqrt{t} < 2, \\ M_m &= 0.926\sqrt{t} \text{ for } 2 \leq \sqrt{t} \leq 3.464, \\ M_m &= 3.21 \text{ for } \sqrt{t} > 3.464 \end{aligned}$$

Similarly, M_m for an outside surface flaw is given by:

$$\begin{aligned} M_m &= 1.77 \text{ for } \sqrt{t} < 2, \\ M_m &= 0.893\sqrt{t} \text{ for } 2 \leq \sqrt{t} \leq 3.464, \\ M_m &= 3.09 \text{ for } \sqrt{t} > 3.464 \end{aligned}$$

and p = internal pressure, R_i = vessel inner radius, and t = vessel wall thickness.

For bending stress, the corresponding K_I for the postulated defect is:

$$K_{Ib} = M_b * \text{Maximum Stress, where } M_b \text{ is two-thirds of } M_m$$

The maximum K_I produced by radial thermal gradient for the postulated inside surface defect of G-2120 is $K_{It} = 0.953 \times 10^{-3} \times CR \times t^{2.5}$, where CR is the cooldown rate in $^{\circ}F/hr.$, or for a postulated outside surface defect, $K_{It} = 0.753 \times 10^{-3} \times HU \times t^{2.5}$, where HU is the heatup rate in $^{\circ}F/hr.$

The through-wall temperature difference associated with the maximum thermal K_I can be determined from Fig. G-2214-1. The temperature at any radial distance from the vessel surface can be determined from Fig. G-2214-2 for the maximum thermal K_I .

- (a) The maximum thermal K_I relationship and the temperature relationship in Fig. G-2214-1 are applicable only for the conditions given in G-2214.3(a)(1) and (2).
- (b) Alternatively, the K_I for radial thermal gradient can be calculated for any thermal stress distribution and at any specified time during cooldown for a $1/4$ -thickness inside surface defect using the relationship:

$$K_{It} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3) * \sqrt{\pi a} \quad (4)$$

or similarly, K_{It} during heatup for a $1/4$ -thickness outside surface defect using the relationship:

$$K_{It} = (1.043C_0 + 0.630C_1 + 0.481C_2 + 0.401C_3) * \sqrt{\pi a} \quad (5)$$

where the coefficients C_0 , C_1 , C_2 and C_3 are determined from the thermal stress distribution at

any specified time during the heatup or cooldown using the form:

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3 \quad (6)$$

and x is a variable that represents the radial distance from the appropriate (i.e., inside or outside) surface to any point on the crack front and a is the maximum crack depth.

Note, that equations 3, 4 and 5 were implemented in the OPERLIM computer code, which is the program used to generate the pressure-temperature (P-T) limit curves. No other changes were made to the OPERLIM computer code with regard to P-T calculation methodology. Therefore, the P-T curve methodology is unchanged from that described in WCAP-14040, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves"^[8] Section 2.6 (equations 2.6.2-4 and 2.6.3-1) with the exceptions just described above.

At any time during the heatup or cooldown transient, K_{Ic} is determined by the metal temperature at the tip of a postulated flaw at the 1/4T and 3/4T location, the appropriate value for RT_{NDT} , and the reference fracture toughness curve. The thermal stresses resulting from the temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{It} , for the reference flaw are computed. From Equation 2, the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

For the calculation of the allowable pressure versus coolant temperature during cooldown, the reference flaw of Appendix G to the ASME Code is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on the measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel inner diameter. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the ΔT (temperature) developed during cooldown results in a higher value of K_{Ic} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist so that the increase in K_{Ic} exceeds K_{It} , the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location and, therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and ensures conservative operation of the system for the entire cooldown period.

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the wall. The heatup results in compressive stresses at the inside surface that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{Ic} for the 1/4T crack during heatup is lower than the K_{Ic} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist so that the effects of compressive thermal stresses and lower K_{Ic} values do not offset each other, and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to ensure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of the pressure-temperature limitations for the case in which a 1/4T flaw located at the 1/4T location from the outside surface is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and therefore tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced by constructing a composite curve based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist wherein, over the course of the heatup ramp, the controlling condition switches from the inside to the outside, and the pressure limit must at all times be based on analysis of the most critical criterion.

3.3 Closure Head/Vessel Flange Requirements

10 CFR Part 50, Appendix G addresses the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (3107 psi), which is 621 psig for Catawba Unit 1 reactor vessel.

This requirement was originally based on concerns about the fracture margin in the closure flange region. During the boltup process, stresses in this region typically reach over 70 percent of the steady-state stress without being at steady-state temperature. The margin of 120°F and the pressure limitation of 20 percent of hydrotest pressure were developed using K_{Ia} fracture toughness, in the mid 1970's.

Improved knowledge of fracture toughness and other issues which affect the integrity of the reactor vessel have led to the recent change to allow the use of K_{Ic} in development of pressure-temperature curves, as contained in Code Case N-640, "Alternative Reference Fracture Toughness for Development

of P-T Limit Curves for Section XI, Division 1". The following discussion uses a similar approach (i.e. using K_{Ic}) here to develop equivalent reactor vessel flange temperature requirement.

The geometry of the closure head flange region for a typical Westinghouse four loop plant reactor vessel such as Catawba Unit 1 reactor vessel is shown in Figure 1. The stresses in this region are highest near the outside of the head. Therefore, an outside reference flaw of 25 percent of the wall thickness parallel to the dome to flange weld (i.e. in the direction of the welding) was postulated in this region. To be consistent with ASME Section XI, Appendix G, a safety factor of two was applied and a fracture calculation performed.

Figure 2 shows the crack driving force or stress intensity factor for the postulated flaw in this region, along with a second curve which incorporates the safety factor of two. Note that the stress intensity factor with a safety factor of one for this region does not exceed $55 \text{ ksi}\sqrt{\text{in.}}$, even for postulated flaws up to 50 percent of the wall thickness. For a reference flaw, with a safety factor of two, the applied stress intensity factor is $85.15 \text{ ksi}\sqrt{\text{in.}}$ at 25 percent of the wall thickness.

The determination of the boltup, or reactor vessel flange requirement, is shown in Figure 3, where the fracture toughness is plotted as a function of the temperature. In this figure, the intersection between the stress intensity factor curve and the K_{Ia} toughness curve occurs at a value slightly higher than $T - RT_{NDT} = 100^\circ\text{F}$, which is in the range of the existing 120°F requirement. The reference calculation used for the original requirement (which is no longer available) resulted in a temperature requirement $T - RT_{NDT} = 120^\circ\text{F}$. Note that the use of K_{Ic} curve to determine this requirement results in a revised requirement of $T - RT_{NDT} = 45^\circ\text{F}$, as seen in Figure 3.

Therefore, the appropriate reactor vessel flange temperature requirement for use with the K_{Ic} curve is as follows:

The pressure in the vessel should not exceed 20 percent of the pre-service hydro-test pressure until the temperature exceeds $T - RT_{NDT} = 45^\circ\text{F}$. This requirement has been implemented with the curves presented in this report.

The limiting unirradiated RT_{NDT} of -4°F occurs in the closure head flange of the Catawba Unit 1 reactor vessel, so the minimum allowable temperature of this region is 41°F at pressures greater than 621 psig with no margins for instrument uncertainties. However, the recommended boltup temperature of 60°F is used in development of the P-T Curves herein.

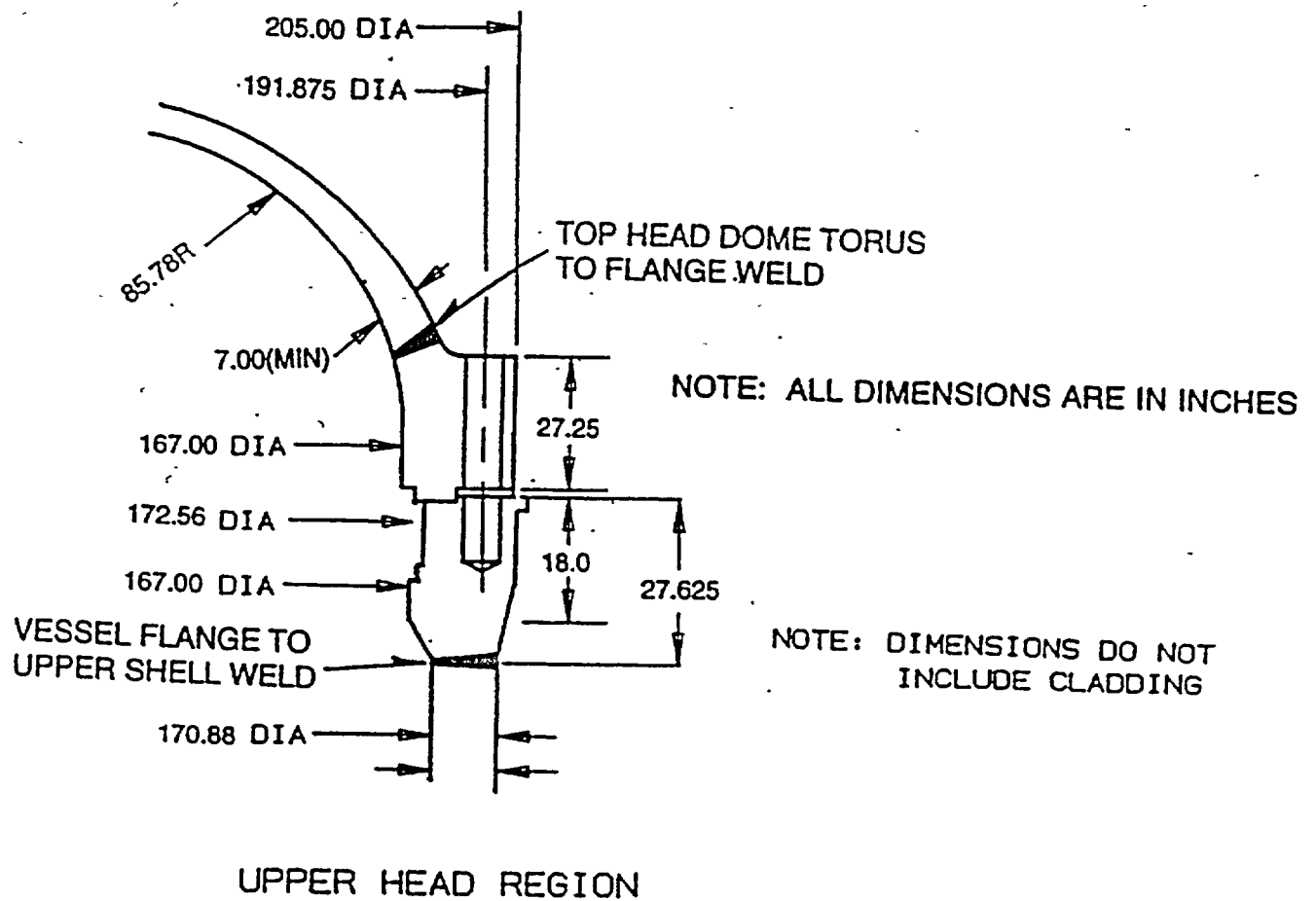


Figure 1 Geometry of the Upper Head/Flange Region of a Typical Westinghouse Four Loop Plant Reactor Vessel

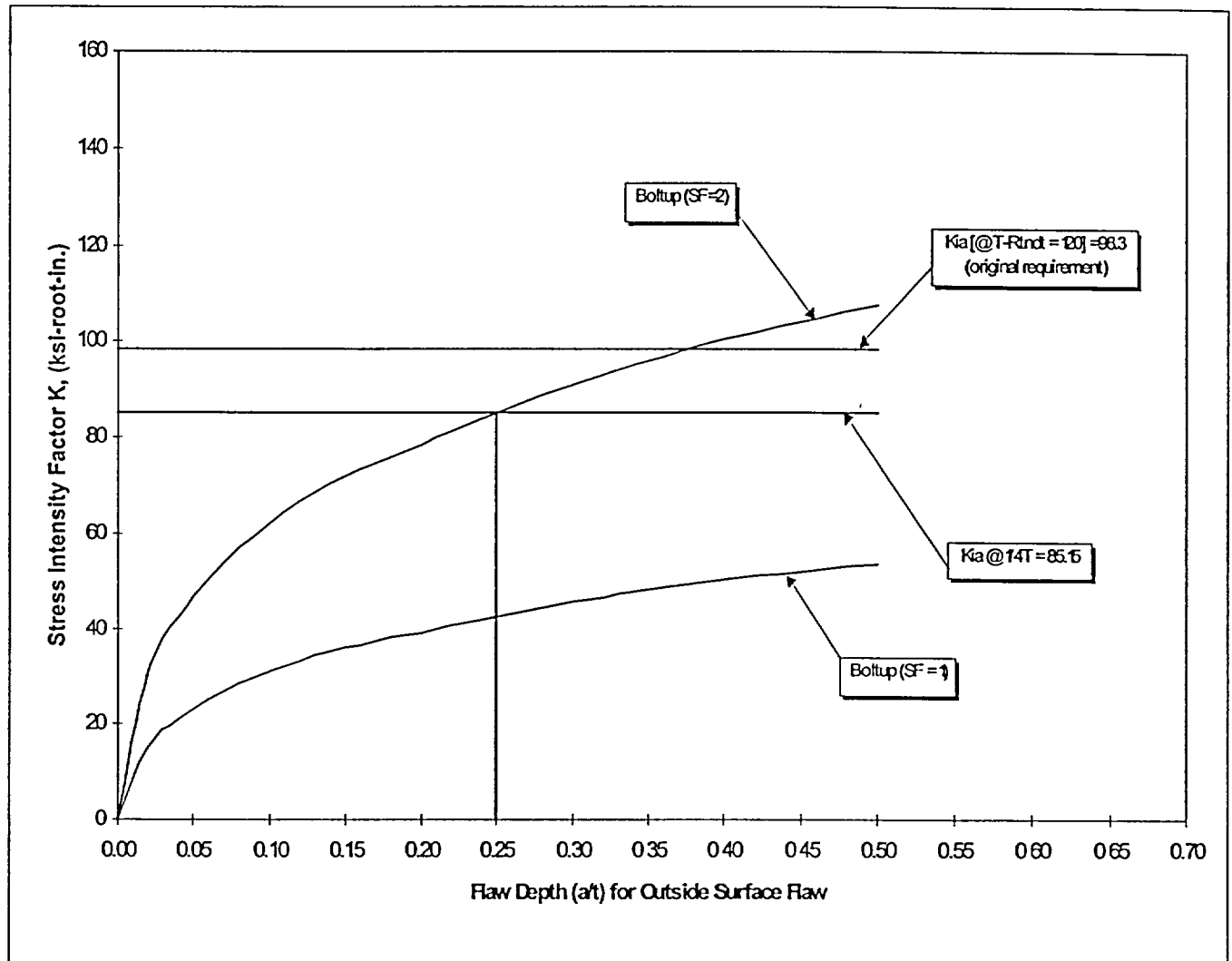


Figure 2 Crack Driving Force as a Function of Flaw Size: Outside Surface Flaw in the Closure Head to Flange Region Weld

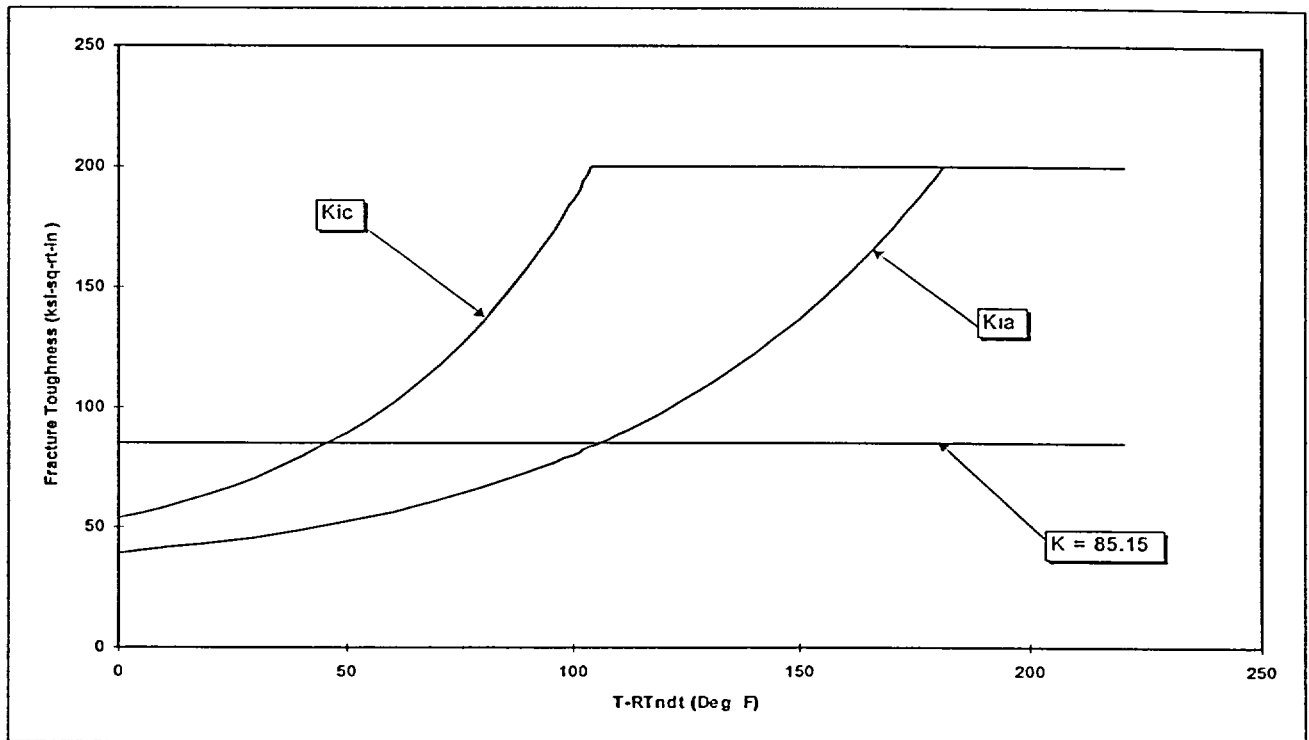


Figure 3 Determination of Boltup Requirement, Using K_{Ic}

4 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

From Regulatory Guide 1.99, Revision 2, the adjusted reference temperature (ART) for each material in the beltline region is given by the following expression:

$$\text{ART} = \text{Initial RT}_{\text{NDT}} + \Delta\text{RT}_{\text{NDT}} + \text{Margin} \quad (7)$$

Initial RT_{NDT} is the reference temperature for the unirradiated material as defined in paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code^[7]. If measured values of initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may be used if there are sufficient test results to establish a mean and standard deviation for the class.

$\Delta\text{RT}_{\text{NDT}}$ is the mean value of the adjustment in reference temperature caused by irradiation and should be calculated as follows:

$$\Delta\text{RT}_{\text{NDT}} = \text{CF} * f^{(0.28 - 0.10 \log f)} \quad (8)$$

To calculate $\Delta\text{RT}_{\text{NDT}}$ at any depth (e.g., at 1/4T or 3/4T), the following formula must first be used to attenuate the fluence at the specific depth.

$$f_{(\text{depth } x)} = f_{\text{surface}} * e^{(-0.24x)} \quad (9)$$

Where x inches (vessel beltline thickness is 8.465 inches) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then placed in Equation 8 to calculate the $\Delta\text{RT}_{\text{NDT}}$ at the specific depth.

The Westinghouse Radiation Engineering and Analysis Group evaluated the vessel fluence projections as a part of WCAP-15117 and are also presented in a condensed version in Table 1 of this report. The evaluation used the ENDF/B-VI scattering cross-section data set. This is consistent with methods presented in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves"^[8]. Table 1 contains the calculated vessel surface fluences values along with the Regulatory Guide 1.99, Revision 2, 1/4T and 3/4T calculated fluences used to calculate the ART values for all beltline materials in the Catawba Unit 1 reactor vessel. Additionally, the surveillance capsule fluence values are presented in Table 2.

TABLE 1
Summary of the Vessel Surface, 1/4T and 3/4T Fluence Values
used for the Generation of the 34 EFPY Heatup/Cooldown Curves

Material	Surface	1/4 T	3/4 T
Intermediate Shell Forging 05	1.98×10^{19}	1.19×10^{19}	4.31×10^{18}
Lower Shell Forging 04	1.98×10^{19}	1.19×10^{19}	4.31×10^{18}
Intermediate to Lower Shell Circumferential Weld Seam	1.98×10^{19}	1.19×10^{19}	4.31×10^{18}

Notes:

- (a) These fluence values were obtained from the calculated fluence values given in Table 6-13 of WCAP-15117 and Table 4-1 in WCAP-15118.
- (b) $1/4T$ and $3/4T = F_{(Surface)} * e^{(-0.24*x)}$, where x is the depth into the vessel wall (i.e. $8.465*0.25$ or 0.75)

TABLE 2*
Integrated Neutron Exposure of the Catawba Unit 1, McGuire Unit 2, and Watts Bar Unit 1
Surveillance Capsules Tested To Date

Plant	Capsule	Fluence
Catawba Unit 1	Z	$2.993 \times 10^{18} \text{ n/cm}^2$, (E > 1.0 MeV)
Catawba Unit 1	Y	$1.318 \times 10^{19} \text{ n/cm}^2$, (E > 1.0 MeV)
Catawba Unit 1	V	$2.334 \times 10^{19} \text{ n/cm}^2$, (E > 1.0 MeV)
Catawba Unit 1	U	$2.439 \times 10^{19} \text{ n/cm}^2$, (E > 1.0 MeV)
Catawba Unit 1	X	$2.439 \times 10^{19} \text{ n/cm}^2$, (E > 1.0 MeV)
McGuire Unit 2	V	$3.268 \times 10^{18} \text{ n/cm}^2$, (E > 1.0 MeV)
McGuire Unit 2	X	$1.406 \times 10^{19} \text{ n/cm}^2$, (E > 1.0 MeV)
McGuire Unit 2	U	$1.962 \times 10^{19} \text{ n/cm}^2$, (E > 1.0 MeV)
McGuire Unit 2	W	$2.969 \times 10^{19} \text{ n/cm}^2$, (E > 1.0 MeV)
Watts Bar Unit 1	U	$5.05 \times 10^{18} \text{ n/cm}^2$, (E > 1.0 MeV)

* This data was taken from Table 4-2 in WCAP-15118.

Margin is calculated as, $M = 2 \sqrt{\sigma_i^2 + \sigma_\Delta^2}$. The standard deviation for the initial RT_{NDT} margin term, is σ_i , 0°F when the initial RT_{NDT} is a measured value, and 17°F when a generic value is available. The standard deviation for the ΔRT_{NDT} margin term, σ_Δ , is 17°F for plates or forgings, and 8.5°F for plates or forgings when surveillance data is used. For welds, σ_Δ is equal to 28°F when surveillance capsule data is not used, and is 14°F (half the value) when credible surveillance capsule data is used. σ_Δ need not exceed 0.5 times the mean value of ΔRT_{NDT} .

Contained in Table 3 is a summary of the Measured 30 ft-lb transition temperature shifts of the beltline materials^[1]. These measured shift values were obtained using CVGRAPH, Version 4.1^[9], which is a hyperbolic tangent curve-fitting program.

TABLE 3
Measured 30 ft-lb Transition Temperature Shifts of all Available Surveillance Data

Material	Capsule	Measured 30 ft-lb Transition Temperature Shift
Intermediate Shell Forging 05 (Tangential Orientation) (Reference WCAP-15117)	Z	-14.9°F ^(a)
	Y	19.09°F
	V	25.61°F
Intermediate Shell Forging 05 (Axial Orientation) (Reference WCAP-15117)	Z	15.74°F
	Y	48.63°F
	V	50.58°F
Catawba Unit 1 Surveillance Weld Metal Data (Reference WCAP-15117)	Z	1.91°F
	Y	17.79°F
	V	26.5°F
McGuire Unit 2 Surveillance Weld Metal Data (Reference WCAP-14799 ⁽¹⁰⁾)	V	38.51°F
	X	35.93°F
	U	23.81°F
	W	43.76°F
Watts Bar Unit 1 Surveillance Weld Metal Data (Reference WCA P-15046 ⁽¹¹⁾)	U	-6.0°F ^(a)

Notes:

(a) This value will be assumed to be 0°F in this evaluation for conservatism (i.e. higher CF value).

Table 4 contains a summary of the weight percent of copper, the weight percent of nickel and the initial RT_{NDT} of the beltline materials and vessel flanges. The weight percent values of Cu and Ni given in Table 4 were used to generate the calculated chemistry factor (CF) values based on Tables 1 and 2 of Regulatory Guide 1.99, Revision 2, and presented in Table 6. Table 5 provides the calculation of the CF values based on surveillance capsule data, Regulatory Guide 1.99, Revision 2, Position 2.1, which are also summarized in Table 6.

TABLE 4^(c)
Reactor Vessel Beltline Material Unirradiated Toughness Properties

Material Description	Cu (%)	Ni(%)	Initial RT _{NDT} ^(b)
Closure Head Flange	0.05	0.83	-4°F
Vessel Flange	--	0.86	-31°F
Intermediate Shell Forging 05	0.09	0.86	-8°F
Lower Shell Forging 04	0.04	0.83	-13°F
Intermediate to Lower Shell Girth Weld Seam ^(b)	0.04	0.72	-51°F
Catawba Unit 1 Surveillance Weld Metal ^(b)	0.05	0.73	--
McGuire Unit 2 Surveillance Weld Metal ^(b)	0.04	0.74	--
Watts Bar Unit 1 Surveillance Weld Metal ^(b)	0.03	0.75	--
Watts Bar Unit 2 Surveillance Weld Metal ^(b)	0.02	0.69	--

Notes:

- (a) Based on measured data. The RVIS Database contains four IRT_{NDT} Values (-51°F, -68°F, -43°F and -50°F) for weld wire heat # 895075. The average of these IRT_{NDT} values is -53°F. However, -51°F is the measured value for the Catawba Unit 1 weld metal. -51°F is used in this evaluation since it is more conservative than the average of -53°F.
- (b) The surveillance weld was made with the same weld wire and flux as the intermediate to lower shell girth weld (weld wire heat # 895075, type Grau L.O. # LW320 flux, Lot #P46).
- (c) This data was taken from Table 4-7 in WCAP-15118.

TABLE 5⁽⁷⁾
Calculation of Chemistry Factors using Catawba Unit 1 Surveillance Capsule Data

Material	Capsule	Capsule $f^{(1)}$	FF ⁽²⁾	$\Delta RT_{NDT}^{(3)}$	FF* ΔRT_{NDT}	FF ²
Intermediate Shell Forging 05 (Tangential)	Z	0.2993	0.670	0 (-14.9)°F ⁽⁵⁾	0°F	0.45
	Y	1.318	1.077	19.09°F	20.56°F	1.16
	V	2.334	1.229	25.61°F	31.47°F	1.51
Intermediate Shell Forging 05 (Axial)	Z	0.2993	0.670	15.74°F	10.55°F	0.45
	Y	1.318	1.077	48.63°F	52.37°F	1.16
	V	2.334	1.229	50.58°F	62.16°F	1.51
	SUM				177.11°F	6.24
	$CF_{\text{Forging 05}} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (177.11) \div (6.24) = 28.38^\circ\text{F}$					
Beltline Region Weld Metal ^(4, 6)	Z (DCP)	0.2993	0.670	1.51°F	1.01°F	0.45
	Y (DCP)	1.318	1.077	14.05°F	15.13°F	1.16
	V (DCP)	2.334	1.229	20.94°F	25.74°F	1.51
	V (DBP)	0.3268	0.692	38.51°F	26.65°F	0.48
	X (DBP)	1.406	1.095	35.93°F	39.34°F	1.20
	U (DBP)	1.962	1.184	23.81°F	28.19°F	1.40
	W (DBP)	2.969	1.288	43.76°F	56.36°F	1.66
	U (WAT)	0.505	0.809	0 (-6.0)°F ⁽⁵⁾	0°F	0.65
	SUM				192.42°F	8.51
$CF_{\text{SP Weld}} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (192.42) \div (8.51) = 22.61^\circ\text{F}$						

Notes:

- (1) f = Integrated neutron fluence from References 1, 10 and 11, ($\times 10^{19}$ n/cm², $E > 1.0$ MeV).
- (2) FF = fluence factor = $f^{(0.28 - 0.1 \cdot \log f)}$
- (3) ΔRT_{NDT} values are measured⁽¹⁾.
- (4) McGuire Unit 2 operates with an inlet temperature of approximately 554°F, Catawba Unit 1 operates with an inlet temperature of approximately 553°F, and Watts Bar Unit 1 operates with an inlet temperature of approximately 560°F. The measured ΔRT_{NDT} values from the McGuire Unit 2 surveillance program were adjusted by adding 1°F to each measured ΔRT_{NDT} and the Watts Bar Unit 1 surveillance program were adjusted by adding 7°F to each measured ΔRT_{NDT} value before applying the ratio procedure. The surveillance weld metal ΔRT_{NDT} values have been adjusted by a ratio factor of:
 0.79 ($CF_{VW} \div CF_{SW} = 54.0 \div 68.0 = 0.79$) for the Catawba Unit 1 data.
 1.0 ($CF_{VW} \div CF_{SW} = 54.0 \div 54.0 = 1.0$) for the McGuire Unit 2 data.
 1.317 ($CF_{VW} \div CF_{SW} = 54.0 \div 41.0 = 1.317$) for the Watts Bar Unit 1 data.
- (5) Assumed to be 0°F for conservatism (i.e. results in a higher CF).
- (6) DCP = Catawba Unit 1 (Data is from WCAP-15117).
 DBP = McGuire Unit 2 (Data is from WCAP-14799)
 WAT = Watts Bar Unit 1 (Data is from WCAP-15046)
- (7) This data was taken from Table 4-8 in WCAP-15118.

TABLE 6*

Summary of the Catawba Unit 1 Reactor Vessel Beltline Material Chemistry Factors

Material	Reg. Guide 1.99, Rev. 2 Position 1.1 CF's	Reg. Guide 1.99, Rev. 2 Position 2.1 CF's
Intermediate Shell Forging 05	58.0°F	28.4°F
Lower Shell Forging 04	26.0°F	**
Beltline Region Weld Metal	54.0°F	22.6°F

* This data was taken from Table 4-9 in WCAP-15118.

** No surveillance material for forging 04, thus Position 2.1 does not apply.

Contained in Table 7 is a summary of the fluence factors (FF) used in the calculation of adjusted reference temperatures for the Catawba Unit 1 reactor vessel beltline materials.

TABLE 7*

Summary of the Calculated Fluence Factors Used for the Generation of the 34 EFPY Heatup and Cooldown Curves

EFPY	1/4T FF	3/4T FF
34	1.05	0.77

* This data was taken from Table 4-10 in WCAP-15118.

Based on the surveillance program credibility evaluation presented in Appendix D to WCAP-15117, the Catawba Unit 1 surveillance program data is credible. In addition, the surveillance program weld metal is representative of all of the beltline region girth weld seam. Hence, the adjusted reference temperature (ART) must be calculated for 34 EFPY for each beltline material at the 1/4T and 3/4T locations. In addition, ART values must be calculated per Regulatory Guide 1.99, Revision 2, Position 1.1 and 2.1

Contained in Table 8 and 9 are the calculations of the 34 EFPY ART values used for generation of the heatup and cooldown curves. The data contained in table 8 and 9 was taken from Table 4-12 in WCAP-15118.

TABLE 8
Calculation of the ART Values for the 1/4T Location @ 34 EFPY

Material	RG 1.99, R2 Method	CF (°F)	FF	$\Delta RT_{NDT}^{(3)}$ (°F)	Margin ⁽⁴⁾ (°F)	$IRT_{NDT}^{(1)}$ (°F)	ART ⁽²⁾ (°F)
Intermediate Shell Forging 05	Position 1.1	58.0	1.05	60.9	34.0	-8	87
	Position 2.1	28.4	1.05	29.8	17.0	-8	39
Lower Shell Forging 04	Position 1.1	26.0	1.05	27.3	27.3	-13	42
Intermediate to Lower Shell Circumferential Weld Seam	Position 1.1	54.0	1.05	56.7	56.0	-51	62
	Position 2.1	22.6	1.05	23.7	23.7	-51	-4

Notes:

- (1) Initial RT_{NDT} values measured values.
 (2) $ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin (°F)}$
 (3) $\Delta RT_{NDT} = CF * FF$
 (4) $M = 2 * (\sigma_i^2 + \sigma_\Delta^2)^{1/2}$

TABLE 9
Calculation of the ART Values for the 3/4T Location @ 34 EFPY

Material	RG 1.99, R2 Method	CF (°F)	FF	$\Delta RT_{NDT}^{(3)}$ (°F)	Margin ⁽⁴⁾ (°F)	$IRT_{NDT}^{(1)}$ (°F)	ART ⁽²⁾ (°F)
Intermediate Shell Forging 05	Position 1.1	58.0	0.77	44.7	34.0	-8	71
	Position 2.1	28.4	0.77	21.9	17.0	-8	31
Lower Shell Forging 04	Position 1.1	26.0	0.77	20.0	20.0	-13	27
Intermediate to Lower Shell Circumferential Weld Seam	Position 1.1	54.0	0.77	41.6	41.6	-51	32
	Position 2.1	22.6	0.77	17.4	17.4	-51	-16

Notes:

- (1) Initial RT_{NDT} values measured values.
 (2) $ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin (°F)}$
 (3) $\Delta RT_{NDT} = CF * FF$
 (4) $M = 2 * (\sigma_i^2 + \sigma_\Delta^2)^{1/2}$

The intermediate shell forging 05 is the limiting beltline material for the 3/4T case and the lower shell forging 04 is the limiting beltline material for the 1/4T case (See Tables 8 and 9). Contained in Table 10 is a summary of the limiting ARTs to be used in the generation of the Catawba Unit 1 reactor vessel heatup and cooldown curves.

TABLE 10
Summary of the Limiting ART Values Used in the
Generation of the Catawba Unit 1 Heatup/Cooldown Curves

EFPY	1/4T Limiting ART	3/4T Limiting ART
34	42°F	31°F

5 HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES

Pressure-temperature limit curves for normal heatup and cooldown of the primary reactor coolant system have been calculated for the pressure and temperature in the reactor vessel beltline region using the methods^[8, 12] discussed in Sections 3.0 and 4.0 of this report. The pressure difference between the wide-range pressure transmitter and the limiting beltline region has not been accounted for in the pressure-temperature limit curves generated for normal operation.

Figures 4 through 6 present the heatup curves without margins for possible instrumentation errors using heatup rates of 60, 80 and 100°F/hr applicable for the first 34 EFPY. Figure 7 presents the cooldown curves without margins for possible instrumentation errors using cooldown rates of 0, 20, 40, 60 and 100°F/hr applicable for 34 EFPY. Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in Figures 4 through 7. This is in addition to other criteria which must be met before the reactor is made critical, as discussed below in the following paragraphs.

The reactor must not be made critical until pressure-temperature combinations are to the right of the criticality limit line shown in Figures 4 through 6. The straight-line portion of the criticality limit is at the minimum permissible temperature for the 2485 psig inservice hydrostatic test as required by Appendix G to 10 CFR Part 50. The governing equation for the hydrostatic test is defined in Code Case N-640^[3] (approved in February 1999) as follows:

$$1.5 K_{Im} < K_{Ic}$$

where,

K_{Im} is the stress intensity factor covered by membrane (pressure) stress,

$$K_{Ic} = 33.2 + 20.734 e^{[0.02 (T - RT_{NDT})]},$$

T is the minimum permissible metal temperature, and

RT_{NDT} is the metal reference nil-ductility temperature.

The criticality limit curve specifies pressure-temperature limits for core operation to provide additional margin during actual power production as specified in Reference 5. The pressure-temperature limits for core operation (except for low power physics tests) are that the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum permissible temperature in the corresponding pressure-temperature curve for heatup and cooldown calculated as described in Section 3.0 of this report. For the heatup and cooldown curves without margins for instrumentation errors, the minimum temperature for the in service hydrostatic leak tests for the Catawba Unit 1 reactor vessel at 34 EFPY is 103°F. The vertical line drawn from these points on the pressure-temperature curve, intersecting a curve 40°F higher than the pressure-temperature limit curve, constitutes the limit for core operation for the reactor vessel.

Figures 4 through 7 define all of the above limits for ensuring prevention of nonductile failure for the Catawba Unit 1 reactor vessel.

The data points used for the heatup and cooldown pressure-temperature limit curves shown in Figures 4 through 7 are presented in Tables 11 through 14. As seen by comparing these results to that from Tables 5-5 through 5-8 of WCAP-15118, there is a minimum increase in pressure of 520 psig (@ lowest temperature) when K_{Ic} and a relaxed reactor vessel flange requirement is used in the calculation of heatup and cooldown limit curves. This increase in allowable pressure associated with the K_{Ic} methodology and relaxed reactor vessel flange requirement has created a "knee" in the heatup curves which show the transition of the heatup curve from steady-state limiting to heatup rate "X°F/hr.". This can also be seen in Tables 11 through 14.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE & LOWER SHELL FORGING 05 & 04

LIMITING ART VALUES AT 34 EFPY: 1/4T, 42°F

3/4T, 31°F

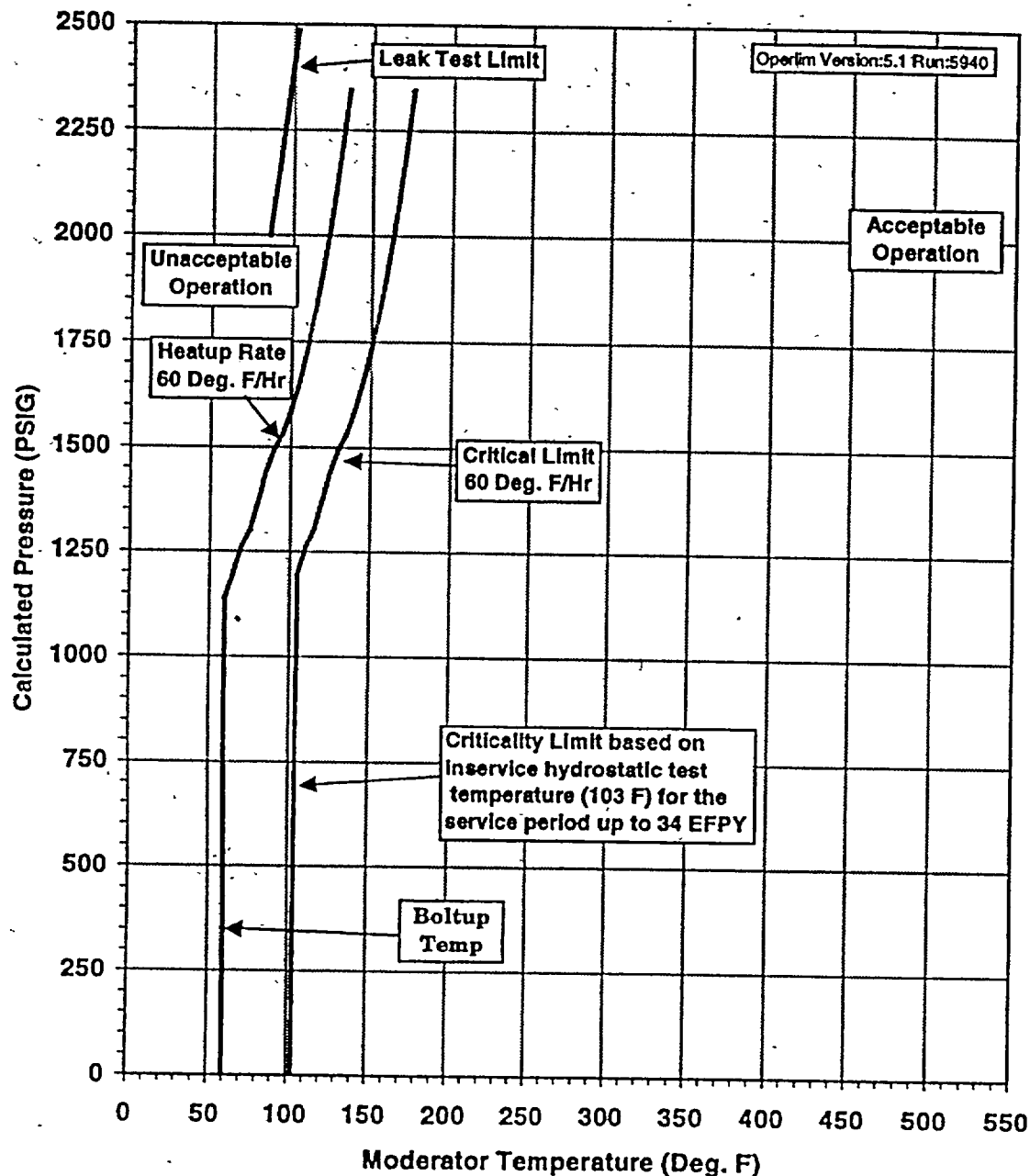


Figure 4 Catawba Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 60°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE & LOWER SHELL FORGING 05 & 04

LIMITING ART VALUES AT 34 EFPY: 1/4T, 42°F

3/4T, 31°F

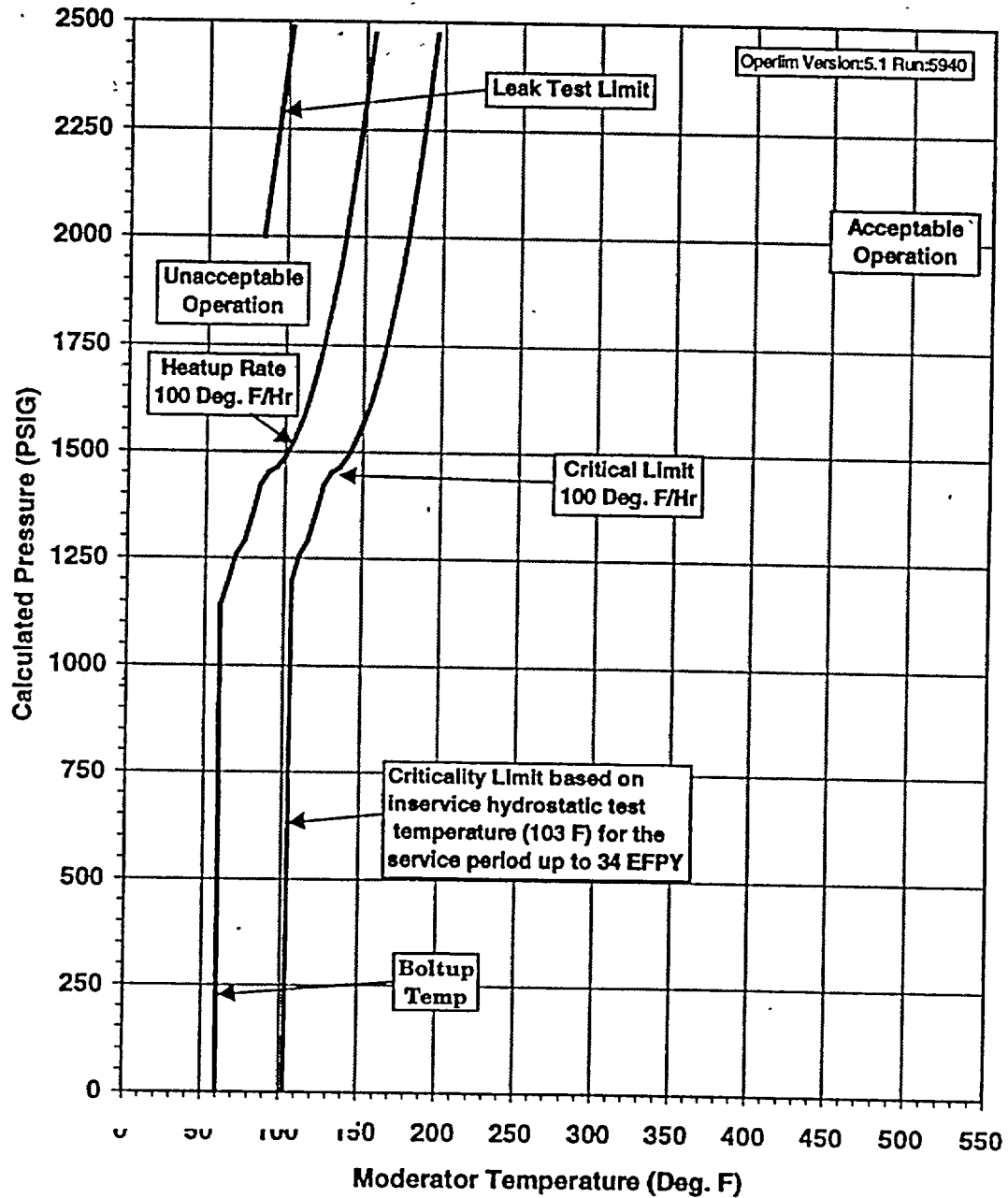


Figure 6 Catawba Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE & LOWER SHELL FORGING 05 & 04

LIMITING ART VALUES AT 34 EFY: 1/4T, 42°F

3/4T, 31°F

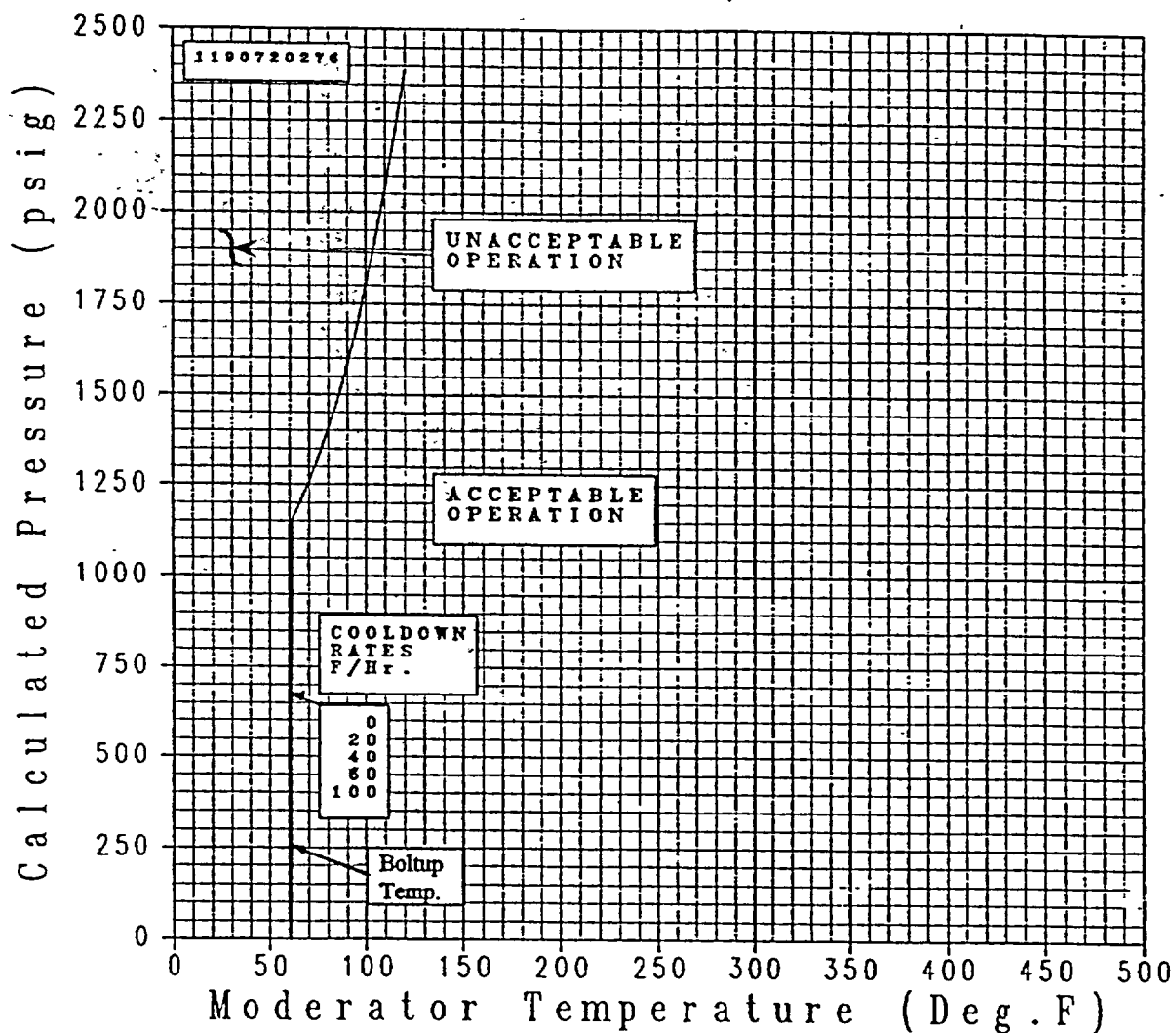


Figure 7 Catawba Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 34 EFY (Without Margins for Instrumentation Errors)

TABLE 11
34 EFPY 60°F/hr. Heatup Curve Data Points Using 1996 App. G
(without Uncertainties for Instrumentation Errors)

Heatup Curves		Configuration #: 1190720276			
60 Heatup		Critical. Limit		Leak Test Limit	
T	P	T	P	T	P
60	0	103	0	86	2000
60	1141	105	1197	103	2485
65	1197	125	1467		
70	1260	130	1493		
75	1329	135	1532		
80	1406	140	1585		
85	1467	145	1652		
90	1493	150	1731		
95	1532	155	1824		
100	1585	160	1931		
105	1652	165	2053		
110	1731	170	2190		
115	1824	175	2344		
120	1931				
125	2053				
130	2190				
135	2344				

TABLE 12
34 EFPY 80°F/hr. Heatup Curve Data Points Using 1996 App. G
(without Uncertainties for Instrumentation Errors)

Heatup Curves		Configuration #: 852799847			
80 Heatup		Critical. Limit		Leak Test Limit	
T	P	T	P	T	P
60	0	103	0	86	2000
60	1141	105	1197	103	2485
65	1197	125	1453		
70	1260	130	1466		
75	1329	135	1490		
80	1406	140	1526		
85	1453	145	1573		
90	1466	150	1631		
95	1490	155	1701		
100	1526	160	1783		
105	1573	165	1878		
110	1631	170	1986		
115	1701	175	2108		
120	1783	180	2245		
125	1878	185	2399		
130	1986				
135	2108				
140	2245				
145	2399				

TABLE 13
34 EFPY 100°F/hr. Heatup Curve Data Points Using 1996 App. G
(without Uncertainties for Instrumentation Errors)

Heatup Curves		Configuration #: 289511205			
100 Heatup		Critical. Limit		Leak Test Limit	
T	P	T	P	T	P
60	0	103	0	86	2000
60	1141	105	1197	103	2485
65	1197	125	1446		
70	1260	130	1450		
75	1329	135	1464		
80	1406	140	1488		
85	1446	145	1522		
90	1450	150	1565		
95	1464	155	1618		
100	1488	160	1682		
105	1522	165	1756		
110	1565	170	1842		
115	1618	175	1940		
120	1682	180	2051		
125	1756	185	2175		
130	1842	190	2315		
135	1940	195	2471		
140	2051				
145	2175				
150	2315				
155	2471				

TABLE 14
34 EFPY Cooldown Curve Data Points Using 1996 App. G
(without Uncertainties for Instrumentation Errors)

Cooldown Curves		Configuration #: 1190720276							
Steady State		20F		40F		60F		100F	
T	P	T	P	T	P	T	P	T	P
60	0	60	0	60	0	60	0	60	0
60	1141	60	1141	60	1141	60	1141	60	1141
65	1197	65	1197	65	1197	65	1197	65	1197
70	1260	70	1260	70	1260	70	1260	70	1260
75	1329	75	1329	75	1329	75	1329	75	1329
80	1406	80	1406	80	1406	80	1406	80	1406
85	1490	85	1490	85	1490	85	1490	85	1490
90	1583	90	1583	90	1583	90	1583	90	1583
95	1687	95	1687	95	1687	95	1687	95	1687
100	1801	100	1801	100	1801	100	1801	100	1801
105	1927	105	1927	105	1927	105	1927	105	1927
110	2066	110	2066	110	2066	110	2066	110	2066
115	2220	115	2220	115	2220	115	2220	115	2220
120	2391	120	2391	120	2391	120	2391	120	2391

6 REFERENCES

1. WCAP-15117, "Analysis of Capsule V and the Dosimeters from Capsules U and X from the Duke Power Company Catawba Unit 1 Reactor Vessel Radiation Surveillance Program," Ed Terek, et al., Dated October 1998.
2. WCAP-15118, "Catawba Unit 1 Heatup and Cooldown Limit Curves For Normal Operation", Ed Terek, Dated October, 1998.
3. ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1", February 26, 1999.
4. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, May 1988.
5. Code of Federal Regulations, 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Washington, D.C., Federal Register, Volume 60, No. 243, dated December 19, 1995.
6. Section XI of the ASME Boiler and Pressure Vessel Code, Appendix G, "Fracture Toughness Criteria for Protection Against Failure.", Dated 1989 & December 1995.
7. 1989 Section III, Division 1 of the ASME Boiler and Pressure Vessel Code, Paragraph NB-2331, "Material for Vessels."
8. WCAP-14040-NP-A, Revision 2, "Methodology used to Develop Cold Overpressure Mitigating system Setpoints and RCS Heatup and Cooldown Limit Curves", J.D. Andrachek, et. al., January 1996.
9. CVGRAPH, Hyperbolic Tangent Curve-Fitting Program, Version 4.1, developed by ATI Consulting, March 1999.
10. WCAP-14799, "Analysis of Capsule W from the Duke Power Company McGuire Unit 2 Reactor Vessel Radiation Surveillance Program", E. Terek, et. al., March 1998.
11. WCAP-15046, "Analysis of Capsule U from the Tennessee Valley Authority Watts Bar Unit 1 Reactor Vessel Radiation Surveillance Program", T.J. Laubham, et. al., June 1998.
12. WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves," W. S. Hazelton, et al., April 1975.

APPENDIX A

TECHNICAL BASIS FOR REDUCED REACTOR VESSEL FLANGE REQUIREMENT

Closure Head/Vessel Flange Requirements: Westinghouse Plants

Introduction

10 CFR Part 50, Appendix G contains the requirements for the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the pre-service hydrostatic test pressure (3107 psig), which is 621 psig for a typical PWR.

This requirement was originally based on concerns about the fracture margin in the closure flange region. During the boltup process, outside surface stresses in this region typically reach over 70 percent of the steady state stress, without being at steady state temperature. The margin of 120°F and the pressure limitation of 20 percent of hydrotest pressure were developed using the K_{Ia} fracture toughness, in the mid 1970s.

Improved knowledge of fracture toughness and other issues which affect the integrity of the reactor vessel have led to the recent change to allow the use of K_{Ic} in the development of pressure-temperature curves, as contained in Code Case N640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1". The following discussion uses a similar approach (i.e., using K_{Ic}) to provide a technical basis for elimination of these flange requirements.

Comparing Flange Requirements

The geometry of the closure head flange region for a typical Westinghouse four loop plant reactor vessel is shown in Figure 1. This geometry was chosen as it is the governing case for all Westinghouse plant designs. The stresses in this region are highest near the outside surface of the head. Hence, a outside reference flaw of 25 percent of the wall thickness parallel to the dome to flange weld (i.e., in the direction of welding) was postulated in this region. All of the other plant designs have a smaller head thickness, so the boltup stresses will be lower, and this case will conservatively bracket the others. To be consistent with ASME Section XI, Appendix G, a safety factor of two was applied for the fracture calculation.

Figure 2 shows the crack driving force or stress intensity factor for the postulated flaw in this region, along with a second curve which incorporates the safety factor of two. Note that the stress intensity factor with a safety factor of one for this region does not exceed $55 \text{ ksi} \sqrt{\text{in}}$, even for postulated flaws of over 50 percent of the wall thickness. For the reference flaw, with the safety factor of two, the applied stress intensity factor is $85.15 \text{ ksi} \sqrt{\text{in}}$ at 25 percent of the wall thickness. The appropriate result for a three loop plant is $70.4 \text{ ksi} \sqrt{\text{in}}$ for the same case, as shown in Figure 4. Since the head thickness for a three loop plant is only 5.75 in., the stresses are lower, resulting in the lower applied stress intensity factor. For two loop plants the geometry is similar and the head thickness is 5.5 inches, so the three loop results apply conservatively to the two loop plants.

Alternative Flange Requirements

The determination of the boltup, or flange requirement, is shown in Figure 3 for the four loop plant, where the fracture toughness is plotted as a function of the temperature. In this figure, the intersection between the stress intensity factor curve and the K_{Ic} toughness curve occurs at a value slightly higher than $T - RT_{NDT} = 45^\circ\text{F}$, which is significantly lower than the existing 120°F requirement, which was originally determined using the K_{Ia} toughness curve. Again the value of $T - RT_{NDT}$ for a three loop plant is even lower, at 29°F , as shown in Figure 5.

The second consideration for the flange region is the pressure at which the requirement is implemented. The existing requirement (20 percent of pre-service hydrotest pressure) was set using the K_{Ia} curve, so an alternative pressure will be developed to maintain the existing margins, using K_{Ic} , to be consistent with the other portions of the pressure-temperature curve development. At $T - RT_{NDT} = 45^\circ\text{F}$, the ratio of $K_{Ic}/K_{Ia} = 85.2/50.9 = 1.67$, so the pressure limitation can be increased to 33 percent of the hydrotest pressure, or 1037 psi, while maintaining the same margins as the existing requirement. The equivalent value for a three loop plant would be 31 percent (950 psi), based on a ratio of $K_{Ic}/K_{Ia} = 1.53$ at $T - RT_{NDT} = 29^\circ\text{F}$.

Are Flange Requirements Necessary?

Therefore we see that using the K_{Ic} curve results in significantly relaxed temperature and pressure for implementing the flange requirement, if we were to change the flange requirements. The key question to be answered is whether the flange requirement is still necessary now that we are using the more realistic K_{Ic} toughness curve. This question can be addressed by examining the stress intensity factor change for a quarter thickness postulated flaw as the vessel is pressurized after boltup, progressing up to steady state operation.

The stresses at the region of interest are shown in Table 1, for steady state operation. Included here are the stresses at the outside surface, which is the highest stress location for this region, as well as the membrane and bending stresses. Note that the OD stresses as well as the membrane and bending stresses, are very similar for the four designs shown in the table. Table 2 shows a comparison of the boltup and steady state stresses for the same plant designs. Again the results are similar for the designs shown, which bracket all plants in service. Unfortunately there are no comparisons available for three or two loop Westinghouse plants, but they will be conservatively covered by the four loop plant results, as discussed above.

As the vessel is pressurized, the stresses in the closure flange region gradually change from mostly bending stresses to mostly membrane stresses. As a result the stress intensity factor, or driving force, increases for a postulated flaw at the outside surface. Table 3 shows the change in stress intensity factor from boltup to steady state operation. Due to differences in geometry the results are slightly different for the designs shown, but the key conclusion is that the change is small, in all cases less than $13 \text{ ksi} \sqrt{\text{in}}$.

Now we will consider the toughness change between K_{Ia} and K_{Ic} . As seen for example in Figure 3, the difference in toughness depends on the temperature chosen for comparison. The most appropriate comparison is at boltup, since that is the lowest temperature of concern for each plant design. To obtain

this temperature, the calculated stress intensity factor at boltup was obtained (column 2 of Table 3), and a safety factor of two applied. The temperature was then determined from the value of K_{Ic} toughness which matched the stress intensity factor discussed above. The difference in toughness was obtained at that temperature.

In reviewing the results of this comparison, we see that the gain in toughness in going to K_{Ic} for the flange considerations is significantly more than the difference in applied stress intensity factor between boltup and steady state operation. Therefore there is no need for any special considerations for the flange when we are using the K_{Ic} toughness. Even if the comparison is made at $T = RT_{NDT}$ the change in toughness exceeds the change in applied stress intensity factor, as seen in Table 3.

Therefore we may conclude that fracture considerations for the reactor vessel closure flange are no longer necessary.

TABLE 1
AXIAL STRESS COMPARISON
STEADY STATE OPERATION @ 2250 PSI

Plant	OD Stress (ksi)	Membrane Stress (ksi)	Bending Stress (ksi)
W 4 Loop	23.0	15.4	7.6
W 3 Loop	21.5	13.3	8.3
CE	22.7	13.1	9.6
B&W	23.8	16.2	7.6

TABLE 2
STRESS COMPARISON
BOLTUP VS STEADY STATE

Plant	Boltup Membrane (ksi)	Boltup Bending (ksi)	SS Membrane (ksi)	SS Bending (ksi)
W 4 Loop	2.1	14.5	15.4	7.6
W 3 Loop	-	-	13.3	8.3
CE	2.1	21.5	13.1	9.6
B&W	4.95	15.4	16.2	7.6

TABLE 3
COMPARISON OF STRESS INTENSITY FACTOR CHANGE
VS FRACTURE TOUGHNESS GAIN

Plant	K (Boltup)	K (Steady State)	ΔK	K_{IA} to K_{IC} Change at Boltup	K_{IA} to K_{IC} at $T = RT_{NDT}$
W 4 Loop	30.8	42.6	11.8	47.3	14.6
W 3 Loop	-	35.2	-	-	14.6
CE	37.8	49.3	12.7	62.5	14.6
B&W	49.5	46.3	-3.2	106.0	14.6

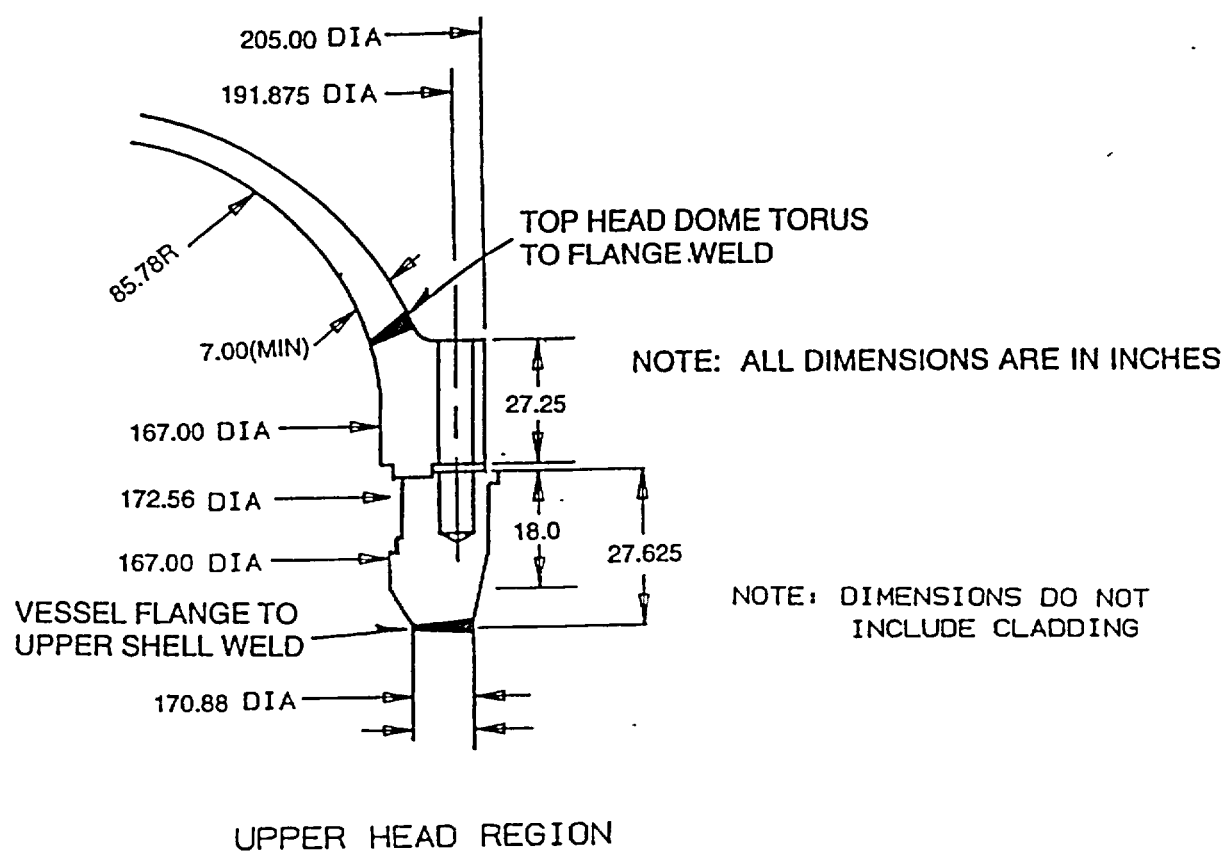


Figure 1. Geometry of the Upper Head/Flange Region of a Typical Westinghouse Four Loop Plant Reactor Vessel

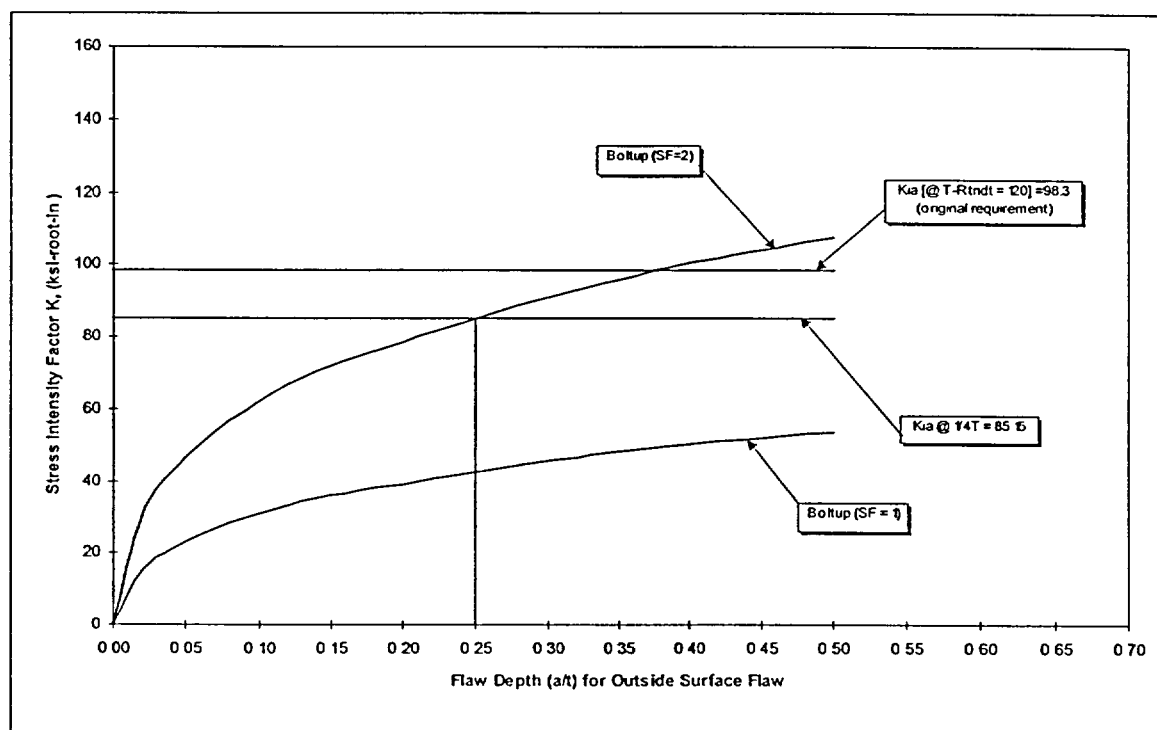


Figure 2. Crack Driving Force as a Function of Flaw Size: Outside Surface Flaw in the Closure Head to Flange Region Weld

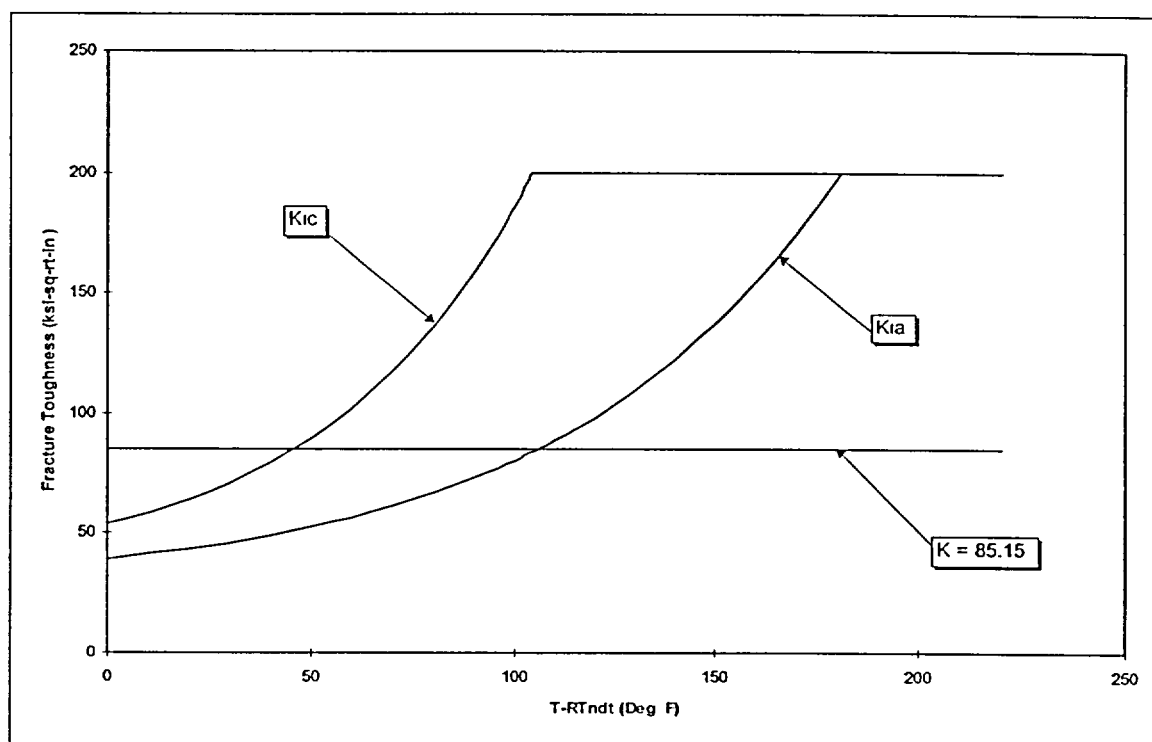


Figure 3. Determination of Boltup Requirement, Four Loop plant, using K_{Ic}

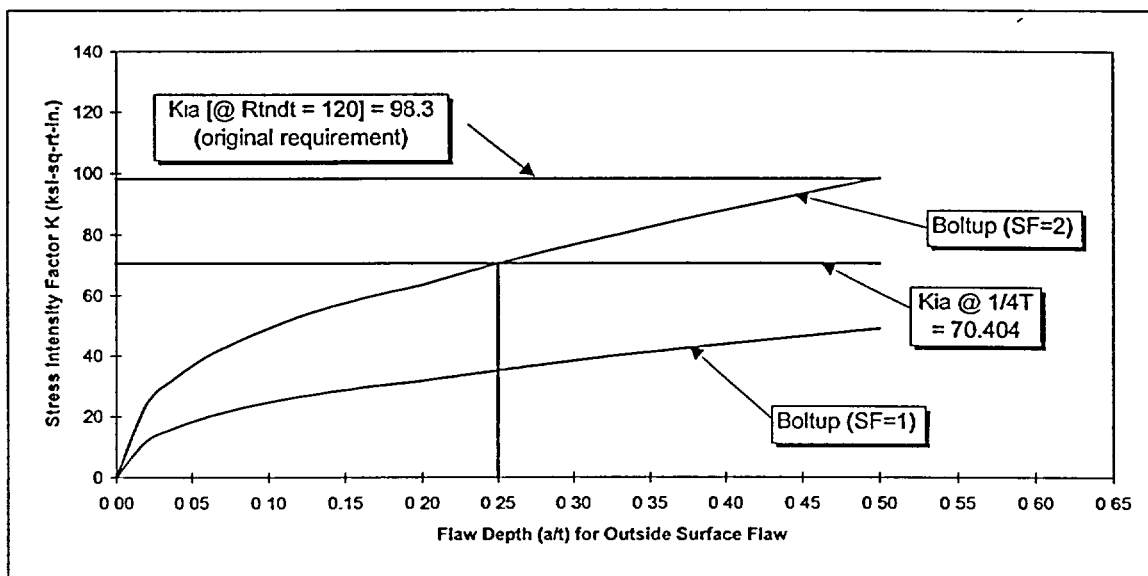


Figure 4. Crack Driving Force as a Function of Flaw Size: Outside Surface Flaw in the Closure Head to Flange Region Weld, Three Loop plant

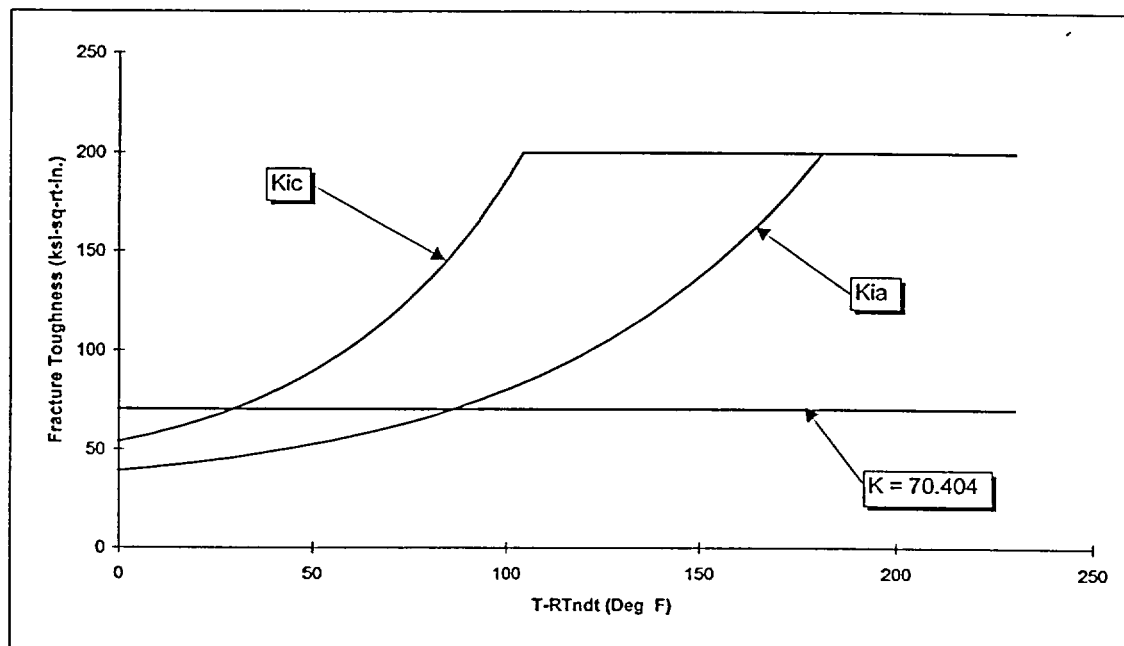


Figure 5. Determination of Boltup Requirement, Three Loop plant, using K_{ic}